U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

15.3.1-15.3.2 LOSS OF FORCED REACTOR COOLANT FLOW INCLUDING TRIP OF PUMP MOTOR AND FLOW CONTROLLER MALFUNCTIONS

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - None

I. AREAS OF REVIEW

A decrease in reactor coolant flow occurring while the plant is at power could result in a degradation of core heat transfer. A resulting increase in fuel temperature and accompanying fuel damage could then result if specified acceptable fuel damage limits are exceeded during the transients. A number of transients that are expected to occur with moderate frequency and that result in a decrease in forced reactor coolant flow rate are covered by this SRP section. Each of these transients should be discussed in individual sections of the applicant's safety analysis report (SAR) as suggested by the Standard Format (Ref. 4).

Core thermal and hydraulic transients associated with partial and complete loss of reactor coolant flow are evaluated. These include:

1. For boiling water reactors (BWRs), partial and complete recirculation pump trips and malfunctions of the recirculation flow controller to cause decreasing flow.

2. For pressurized water reactors (PWR's), partial and complete reactor coolant pump trips.

A partial loss of coolant flow may be caused by a mechanical or electrical failure in a pump motor, a fault in the power supply to the pump motor, a pump motor trip caused by such anomalies as over-current or phase imbalance, or a failure within the recirculation flow control network (BWR) resulting in decreasing flow. A complete loss of forced coolant flow may result from the simultaneous loss of electrical power to all pump motors.

The review includes the postulated initial core and reactor conditions which are pertinent to the loss of flow transient, the methods of thermal and hydraulic analysis, the postulated sequence of events including time delays prior to and after protective system actuation, assumed reactions of reactor systems components, the functional and operational characteristics of the reactor protection system in terms of how it affects the sequence of events, and all operator actions required to secure and maintain the reactor in a safe condition.

The results of the applicant's analyses are reviewed to ensure that values of pertinent system parameters are within expected ranges for the type and class of reactor under review. The system parameters that are evaluated include: core flow and flow distribution, channel heat flux (average and hot), minimum critical heat flux ratio (or minimum critical power ratio), departure from nucleate boiling ratio, vessel water level, thermal power, vessel pressure, steam line pressure (BWR), main steam flow (BWR), and feedwater flow (BWR). The results of the applicant's fuel damage analysis are reviewed by the methods described in SRP Section 4.2 (Ref. 12).

The sequence of events described in the SAR is reviewed by RSB. This review is coordinated with Instrumentation and Control Systems Branch (ICSB). The RSB review concentrates on the need for the reactor protection system; the engineered safety system, and operator action to secure and maintain the reactor in a safe condition.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the RSB reviewer requests initiation of a generic evaluation of the new analytical model by the Core Performance Branch (CPB).

The values of all parameters used in a new analytical model, including the initial conditions of the core and system, are reviewed. It is the responsibility of the RSB reviewer to contact his counterpart in CPB to ensure that the appropriate physics and fuel data have been used in any staff calculations.

In addition, the RSB will coordinate other branches' evaluations that interface with the overall review of the system as follows: The ICSB reviews the instrumentation and control aspects of the sequence described in the SAR to confirm that reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the system analysis as part of its primary review responsibility for SRP Sections 7.2 through 7.5 (Ref. 14 through 17). The CPB, as part of its primary review responsibility for SRP Section 4.4 (Ref. 13), performs generic reviews of the thermal-hydraulic computer models used for this transient and also performs, upon request, additional analyses related to these accidents for selected reactor types. The Procedures Test Review Branch (PTRB) review confirms that a commitment has been made in the SAR to conduct preoperational tests to verify flow coastdown calculations.

For those areas of review identified above as being reviewed as part of the primary review responsibility of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP section of the corresponding review branches.

II. ACCEPTANCE CRITERIA

The RSB acceptance criteria are based on meeting the relevant requirements of the following regulations:

A. General Design Criterion 10 (Ref. 1), as it relates to the reactor coolant system being designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences.

B. General Design Criterion 15 (Ref. 2), as it relates to the reactor coolant system and its associated auxiliaries being designed with appropriate margin to assure that the pressure boundary will not be breeched during normal operations including anticipated operational occurrences.

C. General Design Criterion 26 (Ref. 3) as it relates to the reliable control of reactivity changes to assure that specified acceptable fuel design limits are not exceeded, including anticipated operational occurrences. This is accomplished by assuring that appropriate margin for malfunctions, such as stuck rods, are accounted for.

The basic objectives of the review of loss of forced reactor coolant flow transients are:

1. To identify which of the transients are the most limiting.

2. To verify that, for the most limiting transients, the plant responds to the loss of flow transients in such a way that the criteria regarding fuel damage and system pressure are met.

The specific criteria necessary to meet the relevant requirements of GDC 10, 15 and 26 for incidents of moderate frequency* are:

a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design valves. (Ref. 5)

b. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs and the CPR remains above the MCPR safety limit for BWRs based on acceptable coorelations (see SRP Section 4.4).

c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

d. An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered and is an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. For such accidents, the number of fuel failures must be assumed for all rods for which the DNBR or CPR falls below those values cited above for cladding integrity unless it can be shown, based on an acceptable fuel damage model (see SRP Section 4.2), that fewer failures occur. There shall be no loss of function of any fission product barrier other than the fuel cladding.

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*The term "moderate frequency" is used in this SRP section in the same sense as in the descriptions of design and plant process conditions in References 10 and 11.

The applicant's analysis of the loss of reactor coolant flow transients should use an acceptable analytical model. The equations, sensitivity studies, and models described in References 6 through 9 are acceptable. If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation by CPB.

The values of parameters used in the analytical model would be suitably conservative. The following values are considered acceptable for use in the model:

a. The reactor is initially at rated output (licensed core thermal power) for the number of loops assumed operating, plus 2% to account for power measurement uncertainty, unless a lower power level can be justified by the applicant. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event.

b. Conservative scram characteristics are assumed, i.e., for a PWR maximum time delay with the most reactive rod held out of the core and for a BWR a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate.

c. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

III. REVIEW PROCEDURES

The procedures below are used during both the construction permit (CP) and operating license (OL) reviews. During the CP review the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review state, final values should be used in the analysis, and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The description of each of the loss of reactor coolant flow transients presented by the applicant in the SAR is reviewed by RSB regarding the occurrences leading to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that are required.

5. The extent to which operator actions are required.

6. That appropriate margin for malfunctions, such as stuck rods, are accounted for.

If the SAR states that a particular loss of flow transient is not as limiting as some other similar transients, the reviewer evaluates the justification presented by the applicant. The reviewer confirms that all types of flow loss transients are considered, e.g., pump trips during two-, three-, and four-loop operation. The applicant is to present a quantitative analysis in the SAR of the loss of flow transient that is determined to be most limiting. For this transient, the RSB reviewer, in coordinating with the ICSB reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to adequately limit the consequences of the loss of flow. The RSB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints and evaluates the effects of single active failures of systems and components which may alter the course of the transient. The ICSB review of Chapter 7 of the SAR confirms that the instrumentation and control design is consistent with the requirements for safety systems actions for these events.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam lines are reviewed by the RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, CPB is requested to initiate a generic review of the applicant's proposed model.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by the RSB. Of particular importance are the reactivity coefficients and control rod worths used by the applicant in his analysis, and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated. CPB is consulted regarding the values of the reactivity parameters used in the applicant's analysis.

The results of the analysis are reviewed and compared to the acceptance criteria presented in subsection II of this SRP section regarding the maximum pressure in the reactor coolant and main steam systems. The temporal changes of the neutron power, heat fluxes (average and maximum), reactor coolant system pressure, minimum DNBR (PWR) or CPR (BWR); core and recirculation loop coolant flow rates (BWR), coolant conditions (inlet temperature, core average temperature (PWR), core average steam volume fraction (BWR), average exit and hot channel exit temperatures, and steam fractions), steamline pressure, containment pressure, pressure relief valve flowrate, and flow rate from the reactor coolant system to the containment systems (if applicable) during the transient are reviewed. The important parameters for the loss of reactor coolant flow transients are compared to those predicted for other similar plants to verify that they are within the expected range.

CPB is consulted regarding the specified acceptable fuel design limits (SAFDLs). AEB is notified regarding the extent of fuel failures predicted by the analysis if SAFDLs are exceeded.

The PTRB review confirms that a commitment has been made in the SAR to conduct preoperational tests to verify flow coastdown calculations.

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and his review supports the following kinds of statements and conclusions which should be included in the staff's safety evaluation report (SER):

Several types of plant occurrences can result in an unplanned decrease in reactor coolant flow rate. The ones expected during the life of the plant are those caused by reactor coolant (or recirculation) pump trips or a flow controller malfunction.* All these postulated transients have been reviewed. It was found that the most limiting in regard to core thermal margins and pressure within the reactor coolant and main steam systems was the ________transient. This transient was evaluated by the applicant using a mathematical model that has been reviewed and found acceptable by the staff. The values of the parameters used as input to this model were reviewed and found to be suitably conservative.

The staff concludes that the plant design with regard to transients that are expected to occur during plant life and result in a loss or decrease in forced reactor coolant flow is acceptable and meets the relevant requirements of General Design Criteria 10, 15 and 26. This conclusion is based on the following:

1. The applicant has met the requirements of GDC 10 and 26 with respect to demonstrating that the specified acceptable fuel design limits are not exceeded for this event. This requirement has been met since the results of the analysis showed that the thermal margin limits (MDNBR for PWRs and MCPR for BWRs) are satisfied as indicated by SRP Section 4.4.

2. The applicant has met the requirements of GDC 15 with respect to demonstrating that the reactor coolant pressure boundary limits have not been exceeded for this event. This requirement has been met since the analysis showed that the maximum pressure of the reactor coolant and main steam systems did not exceed 110% of the design pressure.

3. The applicant has met the requirements of GDC 26 with respect to the capability of the reactivity control system to provide adequate control of reactivity during this event while including appropriate margin for stuck rods since the specific acceptable fuel design limits were not exceeded.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

Except in those cases in which the applicant proposed an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

*The SER should present one statement for all similar transients.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 10, "Reactor Design."

2. 10 CFR Part 50, Appendix A, General Design Criterion 15, "Reactor Coolant."

3. 10 CFR Part 50, Appendix A, General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

4. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

5. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

6. "Standard Safety Analysis Report - BWR/6," General Electric Company, April 1973.

7. "Reference Safety Analysis Report - RESAR-3," Westinghouse Nuclear Energy Systems, November 1973. "Reference Safety Analysis Report - RESAR-41," Westinghouse Nuclear Energy Systems, October 1976.

8. "System 80 Standard Safety Analysis Report (CESSAR), " Combustion Engineering, Inc., August 1973.

9. "Standard Nuclear Steam System B-SAR-205, "Babcock & Wilcox Company, February 1976.

10. ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute (1974).

11. ANS Trial Use Standard N212, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," American Nuclear Society (1974).

12. Standard Review Plan Section 4.2, "Fuel System Design."

13. Standard Review Plan Section 4.4, "Thermal and Hydraulic Safety."

14. Standard Review Plan Section 7.2, "Reactor Trip System."

15. Standard Review Plan Section 7.3, "Engineered Safety Features System."

16. Standard Review Plan Section 7.4, "Systems Required for Safe Shutdown."

17. Standard Review Plan Section 7.5, "Safety-Related Display Instrumentation."